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ZIRAT™

Zirconium Alloy Technology Programme

The Annual ZIRAT Programme is focused on fuel assembly material issues and open to nuclear utilities and laboratories. In total, 34 organisations were members in 2024. This programme was started in 1996.

ZIRAT30 Programme

Objective

The overall objective of the ZIRAT Programme is to:

Interpret published R&D results and nuclear plant experience on Zr alloy materials to help the customer to make informed decisions for the improved reliability, safety and economics of operations.

Point out areas where more R&D is needed to resolve potential issues (indications just seen but no problem yet).

The objective is met through review and evaluation of the data on zirconium alloys, identification of the most important new information, and discussion of its significance in relation to fuel performance now and in the future.

The ZIRAT30 Programme consists of Reports and presentations related to the Reports and on other topics not covered in the ZIRAT30 reports.

ZIRAT30 Special Topic Report

The Special Topic Report will cover the range from basic information to current knowledge and be written and explained in such a way that engineers and researchers not familiar with the topic can easily follow the STR, find and grasp the appropriate information. This means that the STR could be used by the organisations in the training of their internal staff with or without the additional assistance of A.N.T. International staff. The background and proposed content of the Report is discussed in detail below.

ZIRAT30 Reports

DHC

Delayed hydride cracking (DHC) has been responsible for cracking in Zr2.5Nb pressure tubes. However, the circumstances for dry storage are quite different from those during residence in the reactor. For dry storage conditions, the existence of a pre-existing, radially-oriented sharp flaw is required; such a flaw can form on the inner surface (such as a PCI crack) or in a hydride rim on the outer surface of the cladding during reactor operation. Such a discontinuity in the cladding wall is required to create a stress gradient that enables hydrogen diffusion to the crack front (tip). The concept of stress intensity factor, denoted as K_I, provides a link between crack depth, a, and applied stress, σ, through $K_I \propto \sigma\sqrt{a}$. DHC is predicted to occur when the value of K_I exceeds a critical value referred to as K_{IH}. Based on the review of the data available in the open literature and assessments or compilations performed by several investigators or organizations, suggested values for K_{IH} applicable to irradiated LWR claddings at temperatures up to ~300 °C were generally found to be ≥ 5 MPa√m.

The overall conclusions are that DHC is a potential cladding degradation mechanism during dry storage, but it is unlikely that the combination of tensile stress and flaw size exceeds the values of K_{IH}. With K_{IH} equal to 5 MPa√m (conservative value) to 3.5 MPa√m (most conservative value) and hoop stress limits of 90 MPa and 120 MPa, shows that the critical flaw size must be between 0.4 to 2.15 times the cladding wall thickness. Critical flaw sizes between 0.4 and nominal wall thickness are highly unlikely and would occur only in rods that would have been heavily damaged during in-reactor operation and still able to withstand, without failing, power transients during reactor operation and shutdown. In addition, past surveillance programs in the US and in Japan have not identified a single rod failure by DHC or by any other mechanism during dry storage. Given the range of potential defects that may exist in the thousands of rods that have been stored and monitored for integrity, the no-failure result indicates that large margins between actual stored spent fuel rod conditions and those required for enabling DHC.

Report Content List

- Introduction
- DHC Mechanism
- Critical stress intensity factor, K_{IH}
- Reviews of publicly available data
- Critical flaw size estimation for DHC of LWR fuel during dry storage
- Conclusions

PCI/PCMI

Pellet-Cladding Interaction (PCI) and Pellet-Cladding Mechanical Interaction (PCMI) may limit fuel operation.

Pellet Cladding Interaction, PCI, is associated with local power ramps during reactor startup or manoeuvring (e.g., rod adjustments/swaps, load following), and occurs under the influence of I, Cs and Cd in a susceptible material that may result in Stress Corrosion Cracking, SCC, of the fuel cladding. The crack always starts at the cladding inner surface and progresses towards the outer cladding diameter in the minute scale. PCI failures may occur in both pressurized water reactors (PWRs)/VVERs and boiling water reactors (BWRs). No PCI defects (failures) will occur provided that: 1) the rod power is below a certain PCI threshold, 2) the power increase step above the PCI threshold or conditioned power level is smaller than a certain amount, and 3) the time above the PCI threshold is short enough.

To prevent PCI failure, it is necessary to remove at least one of the fundamental conditions (tensile stress, sensitive material, aggressive environment), which cause SCC. There are two principal types of remedies:

1) One is to develop reactor operation restrictions that will ensure that the cladding stresses will always be below the PCI threshold stress during power increases and,

2) The second remedy — design improvement — consists of several approaches (fuel cladding design, pellet design, fuel assembly design)

"The ZIRAT meeting exceeded my expectations. The most valuable component of the meeting was the ability to speak directly with the A.N.T. International experts. The depth of knowledge and understanding of all speakers in the A.N.T. network is terrific. The size of the meeting was perfect."

Markus Piro
Canadian Nuclear Laboratories

PCMI (Pellet Cladding Mechanical Interaction), may be a new emerging potential failure mechanism for high burnup fuel. This failure mechanism has however as yet not been seen in commercial BWR/PWR/VVER reactors. The failure mechanism has so far only been seen for high burnup BWR fuel ramped in an experimental reactor. The prerequisite for this type of failure to occur is a hydride layer at the clad outer surface that is cracked during a power ramp. The hydride layer will form an incipient crack that will propagate from the clad to inner surface. The formation of hydride rims at the fuel clad periphery may also decrease the margins towards PCMI failures during RIA events. The margins to PCMI induced RIA failure decrease due to increased concentrated hydride regions with increased burnup. The development of the new more corrosion resistant PWR fuel cladding alloys increases the margin to these failures due to the lower tendency for hydrogen.

The PCI/PCMI topic was addressed in ZIRAT11/IZNA6 and ZIRAT27/IZNA22 Special Topical Reports on PCI and PCMI issued in 2006 [Adamson et al, 2006] and 2023 [Jasiulevicius A.]. The ZIRAT11/IZNA6 report was focused on mechanisms and testing while the ZIRAT27/IZNA22 covered mostly modelling and methodology developments.

This ZIRAT30/IZNA25 Special Topical Report gives an update on all PCI/PCMI aspects covered in the two previously issued Special Topical Reports mentioned above with the following:

Report content list:

- Executive Summary
- Introduction
- Background information on PCI/PCMI
  - o PCI failures in commercial reactors overview from 1960 to current date (including PCI due to Pellet Missing Surfaces)
  - o PCI/PCMI
  - o Prevention of PCI/PCMI failures
    - Fuel design (cladding, pellet, fuel assembly design) changes
  - o Parameters impacting PCI/PCMI
  - o Remedies
    - Liner/Barrier fuel
    - Fuel Lubricants
    - Fuel additives
- PCI/PCMI testing (out-of-pile and in-pile)
- PCI/PCMI modelling and simulation
- PCI/PCMI Operation Limits
- Fuel Vendor Design and Methodologies

Deliverables

A.N.T. International will provide the ZIRAT Members with the following:

- **A Seminar will be held in Spring 2026 in Madrid, Spain to present the results of the ZIRAT30 Programme. The number of full-time employees per Member that may attend the seminar is limited to eight (8) people per organization.**

At the 3 day Seminar, the two ZIRAT30 Special Topic Reports will be presented by the authors. In addition there will be a number of other nuclear fuel related presentations. To be announced later.

- **Before the seminars, the ZIRAT30 members will have access to:**
  - » High-resolution searchable pdf-files with the complete Reports as well as the presentation material in colour.
  - » The files can be copied to a company server, with full read access for everybody with access to the server.
  - » The contents from the Reports and presentation material in pdf-format can be printed. Also, the contents from the pdf-files can be copied and pasted electronically into other documents, e.g. Word files.
  - » All figures and tables with A.N.T International copyright can be used by the member both internally and externally, provided that the source is stated in the caption.

- **The language of the ZIRAT Programme will be English.**

- **The authors will be available for consulting throughout the year. A few telephone or e-mail consultations requiring no additional work are provided at no additional cost to Members.**

- **ZIRAT members have an option to purchase:**
  1. Previous ZIRAT reports at a 50% discount and
  2. Twelve months access to A.N.T. International Online Education Courses at a large discount.
  3. Consulting hours related to large projects for a discounted hourly rate. For more information, click here.
  5. Sign up for access to A.N.T. International Wikipedia (AWIKI), please click here for more information.

Bios of ZIRAT30 Report Authors

Mr. Peter Rudling was a senior consulting scientist at Vattenfall, the largest Swedish power company. Earlier he has also been a Specialist of Fuel Materials at ABB Atom (now Westinghouse) and a Project Manager at EPRI. Peter is now Chairman of the Board of ANT International. Peter has authored/co-authored more than 150 publications as well as 50+ A.N.T. International Reports during 2000-2020. In 2016 Peter got the Kroll Award. The William J. Kroll Zirconium Medal has been established to recognize outstanding achievement in the scientific, technological or commercial aspects of zirconium production and utilization, and to encourage future efforts, studies and research. The award is given to one person per year.

Mr. David Schrire has degrees in Nuclear Engineering from Queen Mary College, London and Purdue University, Indiana. He has worked in the area of fuel performance since 1983, both at Studsvik Nuclear and at ABB Atom (now Westinghouse Sweden) and from 2004 to 2022 at Vattenfall Nuclear Fuel where he was a Senior Specialist in Fuel Performance. He has extensive experience in planning, leading and evaluating in-pile and out-of-pile testing and post-irradiation examination of fuel and core materials, with numerous publications in the area.

While at Studsvik he initiated the ROPE-I and ROPE-II international projects to study the behaviour of high burnup LWR fuel rods operating with a high internal pressure, and later the Studsvik Cladding Integrity Project (SCIP), an international programme to study cladding failure by different mechanisms. At ABB Atom, where he was the head of the fuel inspection and performance group, he led a programme to evaluate the consequences of cladding hydriding. At Vattenfall Nuclear Fuel he was responsible for planning and evaluating the introduction of new fuel designs and materials in the Forsmark and Ringhals reactors, as well as for fuel performance topics including analysis of fuel and fuel structural component failures.

He has been actively involved in international fuel behaviour research programmes including the JAEA ALPS I and II safety tests in Japan and the EPRI Fuel Reliability Programme.

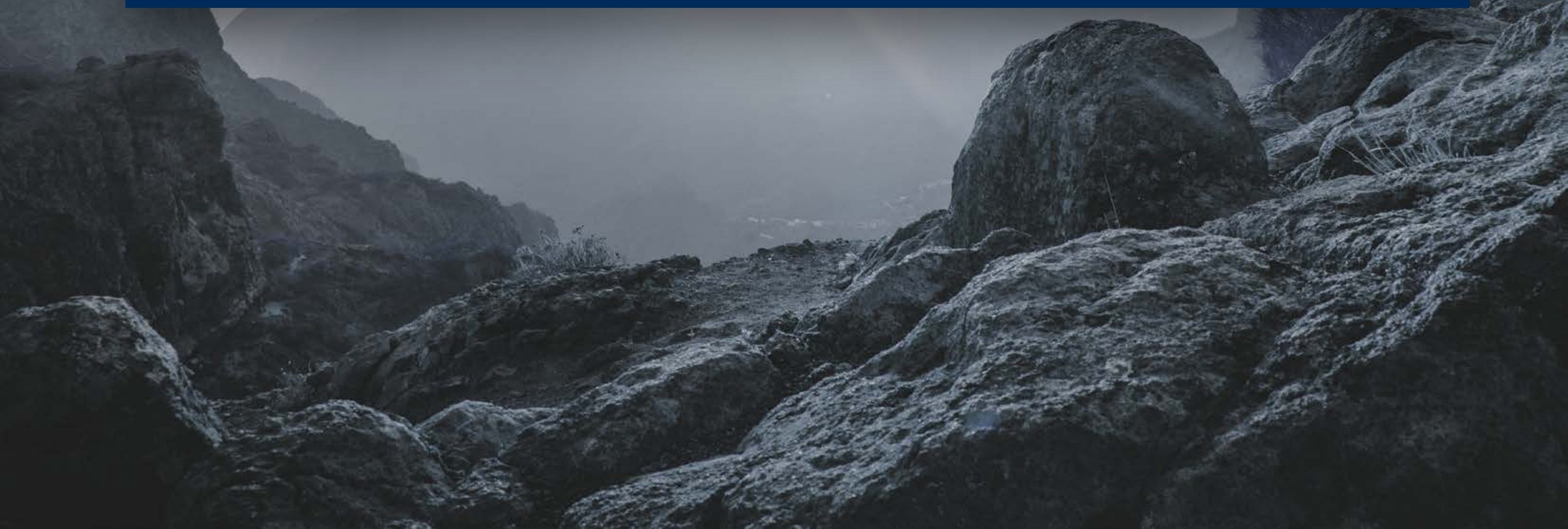
Dr. Sheikh Tahir Mahmood retired from Global Nuclear Fuel in 2012 as a Senior Engineer/Technologist for Fuels Engineering at the Vallecitos Nuclear Center. Earlier he received Masters degrees in Physics and Nuclear Technology from abroad and doctorate in Nuclear Engineering from North Carolina State University. His Post-doctoral work on mechanical anisotropy of zirconium alloys and radiation effects on reactor structural materials was done at NCSU and ORNL, respectively.

At GE Nuclear Energy/GNF, he was actively engaged in fuel performance and materials technology. This activity involved failure root-cause investigations through hot cell PIE of the failed in-core components, and development and evaluation of material property data bases for new materials developed for in-core use. Tahir has particular interest and experience in mechanical metallurgy, mechanical behavior of fuel, cladding and structural materials, and in-reactor behaviour of these materials for improved fuel reliability. He has actively participated in various international nuclear industry research programs.

Dr. Nicolas Waeckel has worked in the Nuclear Engineering Branch of Electricité de France (EDF) from 1984 to 2020. In 2011 he becomes EDF Corporate Expert regarding Fuel & Core safety analysis and related international R&D activities. He spent 3 years at EPRI in Palo Alto in the late 1990 as Project Manager of the Fuel Regulatory Committee of the Fuel Reliability Program. He is Chairman of the international Nuclear Fuel Industry Research program (NFIR) since 2001 and was member of the HALDEN Board (Halden Reactor Project) and of the Studsvik Cladding Integrity Project (SCIP) Board of Management. Nicolas has a PHD and a post-PHD Thesis on Structural Mechanics and heretired from EDF in 2020.

"Excellent presentation material given in a thoroughly professional manner. Excellent interaction with regard to answering the questions raised by the audience."

Christopher Smith
Rolls-Royce



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